September 11, 2000

EA-00-210

Mr. Guy G. Campbell
Vice President - Nuclear
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE - NRC INSPECTION REPORT 50-346/2000007(DRS)

Dear Mr. Campbell:

On July 28, 2000, the NRC completed the first baseline safety system design and performance capability inspection at your Davis-Besse Nuclear Power Station. The results of this inspection were discussed on that day with you and other members of your staff. The enclosed report presents the results of this inspection.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on the design and performance capability of the component cooling water system to ensure it was capable of performing its required post-accident functions.

Based on the results of this inspection, NRC identified seven issues of very low risk significance and two issues for which no risk categorization was assigned. Of these issues, five violations of NRC requirements were established. However, the violations were not cited due to their very low safety significance and because they have been entered into your corrective action program. If you contest these Non-Cited Violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555-001, with copies to the Regional Administrator, Region III, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-001, and the NRC Resident Inspector at the Davis-Besse facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC's Public Document Room **or** from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http:www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any question you have concerning this inspection.

Sincerely,

/RA by Steven A. Reynolds Acting For/

John A. Grobe, Director
Division of Reactor Safety

Docket No. 50-346
License No. NPF-3

Enclosure: Inspection Report 50-346/2000007(DRS)

cc w/encl: B. Saunders, President - FENOC
H. Bergendahl, Plant Manager
D. Lockwood, Manager, Regulatory Affairs
M. O'Reilly, FirstEnergy
State Liaison Officer, State of Ohio
R. Owen, Ohio Department of Health
A. Schriber, Chairman, Ohio Public
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-346
License No:	NPF-3
Report No:	50-346/2000007(DRS)
Licensee:	FirstEnergy Nuclear Operating Company
Facility:	Davis-Besse Nuclear Power Station
Location:	5501 North State Route 2 Oak Harbor, OH 43449-9760
Dates:	July 10 - 28, 2000
Inspectors:	J. Gavula, Team Lead R. Langstaff, Mechanical Inspector D. Schrum, Mechanical Inspector W. Scott, Electrical Inspector O. Mazzoni, Electrical Contractor
Approved by:	John M. Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent little effect on safety. WHITE findings indicate issues with some increased importance to safety, which may require additional NRC inspections. YELLOW findings are more serious issues with an even higher potential to effect safety and would require the NRC to take additional actions. RED findings represent an unacceptable loss of safety margin and would result in the NRC taking significant actions that could include ordering the plant to shut down.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. The color for an indicator corresponds to levels of performance that may result in increased NRC oversight (WHITE), performance that results in definitive, required action by the NRC (YELLOW), and performance that is unacceptable but still provides adequate protection to public health and safety (RED). GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, as described in the matrix. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 50-346/2000007(DRS); FirstEnergy Nuclear Operating Company; Davis-Besse Nuclear Power Station. Safety System Design and Performance Capability.

The inspection was conducted by regional engineering specialists and an outside engineering consultant. The inspection identified seven green issues and two issues with no color, resulting in five Non-Cited Violations. The significance of the issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process.

Cornerstone: Mitigating Systems

- Green. An example of a Non-Cited Violation was identified because the design calculation justifying the degraded grid voltage setpoint had not included all the electrical loads that would start during a design basis accident and it was initially indeterminate whether offsite power could be relied upon under worst case conditions. (Section 1R21.1)

  The risk significance of this issue was determined to be very low because the preliminary results of the revised calculation indicated that the offsite power would be available under worst case conditions.

- Green. An example of a Non-Cited Violation was identified because a design calculation justifying the allowable performance degradation of the component cooling water pumps did not consider the pumps' minimum flow requirements which would have allowed the pumps to operate without sufficient flow.

  The risk significance of this issue was determined to be very low because the current pump performances had not degraded to the extent allowed by the calculation and the licensee had decided to maintain the minimum flow valves in a normally open condition. (Section 1R21.2)

- Green. An example of a Non-Cited Violation was identified because the original design of the component cooling water surge tank had not considered certain failures of unqualified systems which could collapse the tank by causing negative pressures for which it was not specifically designed.

  The risk significance of this issue was determined to be very low because the preliminary results of the licensee's evaluation indicated that the surge tank could withstand a negative pressure beyond a full vacuum. (Section 1R21.3)

- Green. An example of a Non-Cited Violation was identified because the original sizing of the component cooling water surge tank relief valves had not considered the failure of a reactor coolant pressure boundary tube inside the decay heat removal coolers, which could burst the surge tank.

  The risk significance of this issue was determined to be very low because the preliminary results of the licensee's evaluation indicated that the relief valve had sufficient capacity to keep up with the in-leakage from the reactor coolant system and, therefore, would prevent the pressure from exceeding the design limit of the surge tank. (Section 1R21.4)

- Green. An example of a Non-Cited Violation was identified for conducting a test following a modification to the component cooling water pump control circuitry that did not verify that the electronic circuitry affected by the modification would perform as required by the design.

  The risk significance of this issue was determined to be very low because other routine surveillance tests had since been performed which verified that the pump initiating circuits would perform during a design basis accident. (Section 1R21.5)

- Green. An example of a Non-Cited Violation was identified for not having a design basis justifying the use of a leakage acceptance limit in a test for the component cooling water system.

  The risk significance of this issue was determined to be very low because the actual leakage found in the system was extremely low and after doing additional calculations the licensee determined that the make-up system capacity could justify the use of the leakage limit. (Section 1R21.6)

- Green. The licensee failed to verify that the service water system could still provide sufficient make-up to the component cooling water system after more than 20 years without any preventative maintenance. A Non-Cited Violation was identified, in that, there was not an adequate basis to have excluded the component cooling water system from the monitoring requirement of the maintenance rule.

  The risk significance of this issue was determined to be very low because of the extremely remote possibility of the initiating event requiring this function and the high likelihood that an alternate source of make-up could be used. (Section 1R21.7)

- No color. A Non-Cited Violation was identified for no longer complying with the bases of the exemption for the fire protection program requirements within the component cooling water pump room. (Section 1R21.9)

Problem Identification and Resolution

- No color. A Non-Cited Violation was identified for not taking corrective actions to prevent recurrence of the root cause for a modification problem. The modification design had inappropriately installed ground fault protection relays on various pumps, that prevented the pumps from starting or running under normal conditions. (Section 4OA2.1)

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Safety System Design and Performance Capability

Introduction

Inspection of safety system design and performance verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected system to perform its design basis functions. As plants age, their design bases may be lost, such that an important design feature may be altered or disabled. The plant risk assessment model is based on the capability of the as-built safety system to perform its intended safety function successfully. This inspectable area verifies aspects of the mitigating systems and barrier integrity cornerstones for which there are no indicators to measure performance.

The objective of this safety system design and performance capability inspection was to assess the adequacy of calculations, analyses, other engineering documents, and operational and testing practices that were used to support the performance of the component cooling water (CCW) system during normal, abnormal, and accident conditions. The inspection was performed by a team of inspectors that consisted of a team leader, three Region III inspectors, and a contractor.

The CCW system was selected for this inspection, based upon:

- having a high probabilistic risk analysis ranking;
- having had recent significant modifications;
- not having received recent NRC review; and
- providing support to multiple systems.

The criteria used to determine the system's performance included:

- applicable technical specifications;
- applicable Updated Safety Analysis Report (USAR) sections;
- licensee responses and commitments to generic communications; and
- the system design basis document.

a. Inspection Scope

The following system and component inspection attributes were reviewed in detail:

System Needs

Process Medium - water
Energy Source - electrical power
Control System - initiation, control, and shutdown actions
Operator Actions - initiation, monitoring, control, and shutdown

System Condition and Capability

Installed Configuration - elevation and flowpath operation
Design - calculations and procedures
Testing - flowrate, pressure, temperature, voltage, and current

Components

Two components were chosen for detailed review: the CCW pumps and the CCW surge tank. The surge tank is critical in that it is common to both trains. The following attributes were reviewed for both of the chosen components:

Component Degradation
Equipment/Environmental Qualification - temperature (pumps)
Vibration (pumps)
Equipment Protection - flood, missile and freezing (pumps)
Component Inputs and Outputs
Industry Operating Experience

b. Findings

.1 Degraded Grid Voltage Relay Setting

The team was unable to verify the adequacy of the degraded grid voltage relay setpoints of 90 percent of nominal voltage. As described in the licensing basis, following a design basis event, various engineered safety feature motors, including the CCW pumps, will start and the large starting current will depress the electrical bus voltage below 90 percent. A portion of calculation C-EE-004.01-049 was to determine if the voltage would recover above the necessary voltage in time to reset the relays and prevent separation from offsite power. The team's concern with calculation C-EE-004.01-049 was that it assumed SFAS [safety features actuation system] Level 2 actuations instead of SFAS Level 4. The Level 4 SFAS actuations would include additional starting loads and would depress the voltage more than that determined in the calculation. The analysis had to be re-run in order to determine whether recovery from the lower voltage would occur before the built-in time delay timed out. An additional concern was that the switchyard voltage appeared to be routinely lower than that assumed in the calculation.

The licensee issued condition report (CR) 2000-1860 to address these concerns, and was revising the calculation at the end of the inspection. Based on discussions with the licensee following the inspection, the preliminary reanalysis results indicated that equipment operability was not affected and offsite power would be available under worst case conditions. Therefore, this issue was considered to be of very low risk significance (green). However, the failure to include all starting loads in calculation C-EE-004.01.049 was considered an example of a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” in that the adequacy of design was not verified. This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e., green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as CR 2000-1860. (NCV 50-346/2000007-01a)

.2 Pump Minimum Flow Requirements Not Considered in Flow Degradation Calculation

Minimum flow protection for the CCW pumps is provided by normally closed valves that receive open signals based on high differential pressure. With this control method, high differential pressure indicates low flow and low differential pressure indicates high flow. Therefore, pump degradation adversely affects the minimum flow protection by causing the differential pressure switch setpoint to potentially become non-conservative. Calculation C-NSA-016.004-001 determined the allowable CCW pump degradation under worst-case (maximum) flow conditions. However, the calculation did not address the minimum flow protection requirements under degraded conditions. If the pumps were to degrade by 10 percent, as allowed by the calculation, the differential pressure switch setpoint would not have been met and the minimum flow requirement for pumps would not have been satisfied. Low flow conditions can occur under certain alignments and conditions, and extended operation without minimum flow will damage the pumps.

The licensee initiated CR 2000-1851 to evaluate this issue. The most recent inservice tests for all three CCW pumps indicated that the current differential pressure switch setpoint would be reached prior to flows dropping below the minimum flow requirement. In addition, prior to the inspection, the licensee had isolated the control air to the valves in question, such that the minimum flow paths were continually open. Based on the above, this issue was determined to be of very low risk significance (green). However, the failure to address the minimum flow requirements in calculation C-NSA-016.004-001 is considered an example of a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e., green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as CR 2000-1851. (NCV 50-346/2000007-01b)

.3 Surge Tank Not Analyzed for Potential Vacuum Conditions

The CCW surge tank does not have an installed vacuum breaker, but has a cover gas provided by a non-safety related and non-seismically qualified nitrogen supply system. The team identified that a vacuum could develop in the surge tank under design basis earthquake conditions due to the failure of the non-qualified nitrogen supply system and leakage from non-qualified portions of the non-essential headers. The surge tank was not specifically designed to withstand a vacuum and, therefore, the team postulated that it could potentially fail and cause both trains of CCW to be inoperable.

The licensee initiated CR 2000-1781 to address the potential impact of having a vacuum in the surge tank and preliminarily calculated that the tank had a maximum external working pressure of 25.7 pounds per square inch differential. This meant that the tank would not fail even if it were under a complete vacuum; therefore, this issue was considered to be of very low risk significance (green). However, the failure to evaluate the surge tank for potential vacuum conditions was considered an example of a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that the adequacy of design was not verified. This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e. green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 2000-1781. (NCV 50-346/2000007-01c)

.4 Surge Tank Not Analyzed for Pressure Due to Tube Rupture in Decay Heat Cooler

Section 9.2.2.2.3, "System Isolation," of the USAR described the consequences of a rupture (failure) of a tube in a decay heat cooler, which is a reactor coolant system barrier. One of the two outcomes described was that the piping in the CCW system would not rupture. The team questioned whether the surge tank relief valve was capable of relieving the volume of in-leakage from the reactor coolant system for this postulated event. This was potentially critical because the surge tank has the lowest design pressure of all the system's major components, and failure of the surge tank could cause both trains of CCW to be inoperable. The licensee stated that the surge tank relief valve capacity had not considered a rupture of a decay heat cooler tube and did not have any documented bases demonstrating that surge tank relief valve capacity was adequate.

The licensee initiated CR 2000-1844 to evaluate this issue, and had completed a preliminary evaluation of the surge tank relief valve capacity toward the end of the inspection. This evaluation indicated that the relief valve had adequate capacity to prevent the surge tank from exceeding its design pressure rating. Therefore, this issue was considered to be of very low risk significance (green). However, the failure to evaluate the surge tank for a decay heat cooler tube rupture was considered an example of a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," in that the adequacy of design was not verified. This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e., green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 2000-1844.
(NCV 50-346/2000007-01d)

.5 Inadequate Post Modification Testing for Change to CCW Pump Control Logic.

Modification 96-0005 resolved some of the concerns regarding the adequacy of certain fireproofing material by deleting the vulnerable circuits within the CCW pump room. The modification involved the CCW pump control circuitry inside the relay cabinets and the 4160 volt switchgear cubicles. It also deleted several instruments located in the CCW pump room including the low flow and high temperature switches, and the local control stations for CCW pumps 1, 2, and 3. The licensee's specified post-modification testing included only a manual start of the pumps and visual verification of the circuits. The team concluded that the specified post-modification testing was inadequate, in that, it did not include any continuity checks, circuit checks to ground or a functional test to verify that the automatic breaker closure functions operated as designed. After the end of the inspection, the licensee held a telephone conference with team members to provide additional justification for the adequacy of the specified post-modification testing. During these discussions, the team considered that the additional information did not demonstrate the adequacy of the licensee's post-modification testing.

The licensee evaluated this concern and determined that the design basis accident portion of the CCW pump logic had subsequently been demonstrated through the performance of routine surveillance tests DB-SP-03090, 03091, and 03092. Based on the above, this issue was considered to be of very low risk significance (green). However, this was considered an example of a violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, in that, testing required to demonstrate that the pumps would perform satisfactorily in service was not identified as part of the modification. This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e., green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 2000-1852. (NCV 50-346/2000007-02a)

.6 Lack of Basis for Leakage Criteria Specified in CCW System Test

Section 9.2.2.2.4 of the USAR stated that leakage detection and isolation for leakage rates in excess of makeup capacity is provided by a series of alarms and automatic valve closures initiated by level switches on the CCW surge tank. The team reviewed procedure DB-PF-04151, "Augmented CCW Integrated Leakage and Check Valves Leakage Test," to verify that the leakage did not exceed the makeup capacity to ensure that the CCW system would perform satisfactorily in service. The stated purpose of procedure DB-PF-04151 was to verify that system integrity met design requirements with Step 4.1.23 specifying a leakage acceptance criterion of 20 gallons per minute (gpm). When questioned by the team, the licensee was unable to demonstrate that 20 gpm satisfied design requirements, because the makeup capacity to the CCW system had never been previously determined.

During their review of this issue, the licensee initiated a new calculation and determined that the makeup capacity from the service water system was at least 20 gpm. Also, the CCW leakage was less that 0.1 gpm, based on the latest results. Therefore, this issue was considered to be of very low risk significance (green). However, this was considered a violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," in that the test procedure used to demonstrate that the CCW system will perform satisfactorily in service, included acceptance limits that were not contained in applicable design documents. This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e., green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 2000-1843. (NCV 50-346/2000007-02b)

.7 Section 9.2.2.2.4, "Leakage Consideration," of the USAR stated that "Seismic Category I makeup is available from the service water system in the event that non-Seismic I makeup supplies are not available after a Design Basis Accident." The service water makeup capability is provided by 1-inch lines to each train of the CCW system from trains of the service water (SW) system. Both makeup lines have dual isolation provided by two manually operated valves. Although the isolation valves are exercised every two years, flow through the piping has not been verified nor has the piping been cleaned since installation. The team noted that industry experience has shown that lines connected to SW systems, such as these, are susceptible to corrosion and bio-fouling. As such, the team determined that the ability to provide makeup from SW was not being demonstrated. During the inspection, the licensee stated that the function of providing SW makeup to the CCW system was within the scope of the Maintenance Rule (i.e., 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants.") On June 8, 2000, the licensee returned the CCW system to (a)(2) from (a)(1) status of the maintenance rule for monitoring purposes.

10 CFR 50.65(a)(1) requires, in part, that each holder of a license to operate a nuclear power plant to monitor the performance or condition of structures, systems, or components (SSCs) against licensee-established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. 10 CFR 50.65(a)(2) specifies that monitoring as specified by 10 CFR 50.65(a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. As of June 8, 2000, the licensee had not demonstrated that the performance or condition of the SW makeup to CCW system lines had been effectively controlled through the performance of appropriate preventive maintenance such that these lines were capable of performing their intended function. The performance or condition of the SW makeup to CCW system lines had not been demonstrated in that there was no monitoring which provided reasonable assurance that SW makeup (i.e., flow) to the CCW system could be provided. Consequently, the licensee had failed to meet the requirements of 10 CFR 50.65(a)(2) for (a)(2) classification of the CCW system under the Maintenance Rule.

The team determined that a violation of 10 CFR 50.65(a)(2) existed, in that, as of June 8, 2000, the performance or condition of the SW makeup to CCW had not been demonstrated. This violation is associated with an inspection finding that is characterized by the Significance Determination Process as having very low risk significance (i.e. green) because of the remote likelihood of this category of initiating events (i.e, seismic) and the high likelihood of recovering a normal source of makeup.

This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 2000-1779. (NCV 50-346/2000007-03)

.8 CCW Room Sprinklers Partially Obstructed

The team noted that three fire protection system sprinkler heads above the CCW pumps were partially obstructed. During the inspection, the licensee's fire protection engineer stated that obstructions caused by ventilation ducts were acceptable per Section 4-4.13 of NFPA 13-1978, "Standards for the Installation of Sprinkler Systems," because the ventilation ducts were less than four feet wide and did not require an additional sprinkler underneath the duct. The team noted that Section A-4-4.13 of the standard referenced Section B-4-2.3 for cases where ducts were less than four feet wide. Section B-4-2.3, in turn, referenced Section 4-2.4.6 of NFPA 13-1978, which provided specific distance spacing requirements from sprinklers for a given vertical drop. The team was concerned that the existing sprinkler configuration did not appear to have been evaluated against the spacing requirements outlined by Section 4-2.4.6 of the standard. The licensee initiated CR 2000-1858 to evaluate the concern for operability and potential corrective action. The licensee was still evaluating this concern at the end of the inspection. The inspectors noted that even if the obstructions had resulted in moderate degradation of the automatic suppression capability of the fire protection system, the finding would have been characterized by the Significance Determination Process as having very low risk significance (i.e., green) provided no other fire protection related degradations existed in the CCW pump room.

.9 Fire Protection Program Exemption Bases No Longer Met

During walkdowns of the CCW room, the team noted that the three CCW pumps did not meet the Appendix R, Fire Protection Program for Nuclear Power Facilities, separation requirements of 20 feet, and that none of components or circuits had fire protection barriers. The team learned that the licensee had previously requested and was granted an Appendix R exemption by the NRC on November 23, 1982. The exemption stated that items identified in Table 1 (i.e., circuits and components), of their April 29, 1982, Exemption Request, would be protected with a one-hour fire wrap/barrier. Credit was taken for the approved exemption to demonstrate separation or adequacy of fire protection features in the CCW room. However, the CCW room no longer met the exemption bases and, therefore, no longer met Appendix R requirements. The failure to meet Appendix R requirements was considered a violation of 10 CFR Part 50, Appendix R, Section III.G. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1.a of the NRC Enforcement Policy. The licensee entered this issue into their corrective action system as CR 2000-1857.
(NCV 50-346/2000007-04)

The licensee stated that modifications and re-analysis had eliminated the need for protection of any conduits in the CCW pump room; however, they were unable to provide the evaluations needed to support this statement. Without a review, it was not obvious that the current configuration of the CCW Room would be approved even with an amended exemption. This was considered an Unresolved Item until the licensee's evaluations, eliminating the need for fire wraps/barriers within the CCW pump room, can be evaluated. (URI 50-346/2000007-05)

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports and PCAQRs [potential condition adverse to quality reports] associated with component cooling water design issues to verify that the licensee had an appropriate threshold for identifying design issues. The inspectors also evaluated the effectiveness of the corrective actions to the identified issues, including the engineering justification for operability, if applicable.

b. Findings

.1 Inadequate Corrective Action for Modification Design Problem

PCAQRs 98-0775 and 98-0776 addressed a problem with modification 95-0021 for the replacement of several ground fault protection relays that involved CCW pump No. 1 as well as other components. The replacement relays were solid-state relays and had a different detection sensitivity than the original electro-mechanical relays. As a result, the replacement relays tripped the pump breaker when trying to start the pump or under other normal operating conditions. The apparent cause documented in the PCAQRs was “a lack of understanding of the system response of ground current sensing circuits on 4160 volt motor circuits.” The fundamental problem was that the associated motor circuits were producing unanticipated short duration current spikes that the replacement relays could detect whereas the original relays could not. This was considered a significant condition adverse to quality, in that, the associated modification had installed devices causing multiple safety-related motors to be inoperable and this wasn’t identified until after the modification was turned over to operations. The licensee’s corrective actions to prevent recurrence only corrected the immediate problem by adding a time delay to the affected relays. The corrective actions to prevent recurrence did not address the apparent cause documented in the listed PCAQRs. The failure to take corrective action to preclude repetition is considered a violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action.” This Level IV violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as CR 2000-1839.
(NCV 50-346/2000007-06)

.2 Basis for Problem Resolution Questioned

When the above pump-breaker-tripping problem occurred, the licensee’s initial corrective action was to increase the amperage setting of the ground fault protection relays. This decreased the sensitivity of these devices making them potentially less effective. However, after additional problems occurred, it was determined that a time delay should be added to the relays to allow the initial current spikes to not be detected.

This ultimately resolved the problem. In PCAQR 98-0775, Part 5G, "Specific Action Necessary to Correct Condition," stated that the "current settings were increased, but it was not necessary to do this to correct the problem." When questioned by the team, licensee personnel could not provide a basis for why the sensitivity of the ground fault relays had not been returned to the previous sensitivity level. Since the sensitivity had been decreased to resolve a problem and it was then determined that the sensitivity decrease was not needed, the effectiveness of the licensee's problem resolution appeared to be weak. The licensee initiated CR 2000-1859 to review the relay settings.

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. G. Campbell, and other members of licensee management and staff, in an exit meeting on July 28, 2000. The licensee acknowledged the information and findings presented. The inspectors identified the proprietary information reviewed during the inspection and questioned the licensee as to whether proprietary information had been retained. The inspectors also discussed the potential for proprietary information to be included in the inspection report. The licensee confirmed that no proprietary information was retained at the completion of the inspection. The licensee concurred that the proposed inspection report content would not compromise any proprietary information.

.2 Additional Conference Calls with the Licensee

At the request of the licensee, a telephone conference was held on August 16, 2000, to provide the NRC with additional information regarding the post-modification testing issue discussed in paragraph 1R21.5, above. The team's determination regarding the test's adequacy did not change based on the additional information provided by the licensee. The team informed the licensee's regulatory affairs staff of the final characterization of the issues identified by the inspection by telephone conference on September 6, 2000.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Campbell, Vice President, FENOC
D. Geisen, Manager, Technical Services
J. Hartigan, Senior Staff Engineer, DBE
C. Hengge, Senior Engineer, SYME
P. Jacobsen, Senior Engineer, DEEC
G. LeBlanc, Supervisor, DEEC
D. Lockwood, Manager, Regulatory Affairs
D. Miller, Supervisor, Compliance
S. Moffitt, Director, Technical Services
P. Straube, Senior Engineer, DBE
S. Wise, Senior Operator

NRC

S. Reynolds, Deputy Division Director, Region III
J. Jacobson, Branch Chief, Region III
K. Zellers, Senior Resident Inspector, Davis Besse

ITEMS OPENED, CLOSED AND DISCUSSED

Opened

50-346/2000007-05	URI	Verify evaluations eliminating fire wraps in the CCW pump room

Opened and Closed During This Inspection

50-346/2000007-01a	NCV	Failure to include all starting loads in calculation C-EE-004.01.049
50-346/2000007-01b	NCV	Failure to address the min flow in calculation C-NSA-016.004-001
50-346/2000007-01c	NCV	Failure to consider a vacuum in the design of surge tank
50-346/2000007-01d	NCV	Failure to consider DHC tube rupture in design of surge tank
50-346/2000007-02a	NCV	Inadequate post modification testing for Mod 96-0005
50-346/2000007-02b	NCV	Acceptance limits for CCW leakage test did not have a basis
50-346/2000007-03	NCV	Performance of SW makeup to CCW not demonstrated
50-346/2000007-04	NCV	Appendix R exemption bases for CCW pump room no longer met
50-346/2000007-06	NCV	No corrective action to preclude repetition of design problem

Previous Items Closed

None

Previous Items Discussed

None

LIST OF ACRONYMS USED

CCW	Component Cooling Water
CR	Condition Report
DHC	Decay Heat Cooler
gpm	Gallons Per Minute
NCV	Non-Cited Violation
NFPA	National Fire Protection Association
PCAQR	Potential Condition Adverse to Quality Report
SFAS	Safety Features Actuation System
SSC	Structures, Systems, or Components
SW	Service Water
URI	Unresolved Item
USAR	Updated Safety Analysis Report

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings.

<u>Calculations</u>

Bechtel 25-002, CCW Pump Room Area Ventilation
C-EE-0157-004, 4.16kV and 13.8kV Cable Ampacities (Motor Loads), Revision 1, dated 7-18-00
C-EE-004.01-008, Sheet 12, Component Cooling Water Pump 1-3, Time-Current Characteristic Curves for Breaker AD108, dated 10/9/90
C-EE-004.01-005, Protective Relay Setpoint for Component Cooling Water Pump Motor 1-1 (AC113), Revision 1, dated 04/28/98
C-EE-004.01-006, Protective Relay Setpoint for Component Cooling Water Pump Motor 1-2 (AD113), Revision 1, dated 11/23/93
C-EE-004.01-007, Protective Relay Setpoint for Component Cooling Water Pump Motor 1-3 (AC108), Revision 0, dated 12/19/90
C-EE-004.01-049, Page 1, dated 12/20/96 Plant loading
C-EE-004.01-049, Attachment 2, Page 1 of 29, dated 10/23/96
C-ICE-16.03-001, Setpoint for PDSH 3981 and 3982, Revision, 0, 4/13/88
C-ME-016.04-002, CCW Line Overpressurization Review for Mod 88-0145, Revision 0, 6/6/90
C-ME-016.04-029, Relief Valve for CCW Side of the Letdown Coolers, Revision 0, 8/30/99
C-ME-016.05-001, CCW Pump Room Ventilation
C-ME-40.026, CCW Flow for SFAS Levels 3 - 4, Revision 1, 4/23/90
C-NSA-016.04-003, CCW Heat-Up Rate
C-NSA-049, Decay Heat Pump Inboard Bearing Evaluation with Cold CCW During LOCA.
C-NSA-016.02-001, CCW Pump 1-2 Allowable Pump Degradation, Revision 0, 5/11/94
C-NSA-016.04-002, CCW Surge Tank Level Validation for PCAQR 98-1858, Revision 1, 8/20/99
C-NSA-016.04-004, CCW Pump NPSH Requirements, Revision 0, dated 7/19/00
Engineering Evaluation, Effect of a Fire in Room 328 on CCW Pump Availability, 4/27/90
EC0106AX, Synchro. Check Relay Setting, Davis-Besse Unit 1, Revision 0, dated 6/18/85
Mech 40.11, Sizing Basis for Essential Makeup, 1/14/76

TB4578, Davis-Besse Nuclear Power Station Unit 1 Transient Stability Study TECO letter to Bechtel, dated 11/3/78
1999-01079, Relief Valve Sizing for Letdown Cooler Tube Rupture, Revision 1, 8/6/99
Load Summary by Bus, S&L Project No. 1503003, dated 01/06/00

Condition Reports Initiated as a Result of Inspection

2000-1779, Flow Path for SW to CCW Makeup Not Verified
2000-1781, Surge Tank Is Not Designed for Negative Pressure and Could Collapse
2000-1838, Calc C-EE-004-01.049 Requires Enhancement for Plant Loading Design Bases
2000-1839, Corrective action for CR's on MOD 95-0021 didn't address design problem
2000-1843, No Documented Design Basis for Allowable Leakage from the CCW System
2000-1844, Incomplete Design Bases Evaluation of a Decay Heat Cooler Tube Rupture
2000-1851, C-NSA-016.04-001 Didn't Address Min Flow for Degraded Pump Condition
2000-1852, Lack of Functional Testing Following Modification 96-0005
2000-1857, Revision to Appendix R Exemption for CCW Pump Room
2000-1858, Obstructions of the Installed Sprinklers Over the CCW Pumps
2000-1860, Basis for 90% Degraded Grid Voltage Not Accounted for in Relay Setting
2000-1861, Aux. Transformer Backfeed Not Analyzed

Condition Reports/ Potential Condition Adverse to Quality Reports

1991-0427, CC91 and CC531 Failed to Meet Reverse Flow Test Requirements
1997-0048, Service Water Pump Quarterly Test Isolates Portions of the Standby CCW Hx
1997-0375, CCW Pump Not be Powered for 50 Seconds After EDG Start Signal W/o SFAS
1997-0424, CCW Declared Inoperable During Performance of Annual Ventilation Test
1997-0752, CCW System is Leaking 6 Gallons/Hour
1997-0108, No 3 CCW Heat Exchanger Temperature Control Valve Failed During Test
1997-0592, Automatic Valves in CCW Train 1 Loop Not Verified Monthly as Required
1998-0001, Evaluate Outside Ambient Temperatures Above 95 Degrees for the CCW Rooms
1998-0776, Ground Fault Relay 50GS Tripped When Starting of the Service Water Pump
1998-0797, Post Testing for Modification 95-0021
1998-1011, High Pressure Condition in CCW Loop 2
1998-1206, Detent Device on CC 1469 Failed to Retract During Valve Stroke Test
1998-1244, Insulation on Floor in CCW Room
1998-1511, Unable to Meet the Accept Criteria for CCW 1 Heat Exchanger of 57MBTU/Hour
1998-1623, CCW Heat Exchanger 2 Did Not Meet DB-PF-04705 Acceptance Criteria
1998-1643, CCW Train 2 HVAC Outside Air Damper HV5444C Failed to Stroke Full Closed
1998-1674, Incorrect Data Entry for CCW Heat Exchanger Test
1998-1858, CCW Leakage,
1999-0039, Increased Make-Up for the CCW Surge Tank
1999-0196, Letdown Cooler Rupture Disc Leak
1999-1088, CCW Leakage
1999-1412, CCW Elevator Room Exhaust Damper Disabled Due to HELB Concern
1999-1496, CC1495 CCW Non-Essential Isolation Valve Did Not Fully Close
1999-1607, Loose Scaffolding Lying Across CCW Valve CC 5
1999-1686, SW Flow Control Valves to CCW Heat Exchangers Fail Open on Loss of Air
1999-1688, CCW Surge Tank Low-Level Setpoints and Non-Essential Loop Isolation Valve Stroke Times Have Not Been Shown to Prevent Depletion of the CCW Surge Tank

1999-1692, CCW Pump Room Ventilation Cannot Respond to Cooling Needs of the Room
1999-1699, Inadequate Response to Information Notice 89-54, dated 10/07/99
1999-1720, Issue with Adequacy of Past Corrective Actions on CCW Problems
1999-1757, Exhaust Damper Linkage Disconnected for CCW 1 Ventilation
1999-2111, CCW Valve CC1495 Did Not Stroke Fully During Performance Test
1999-2202, CCW Pump Room Temperature
2000-0348, Potential Degradation of Air Regulators for CC1469
2000-1081, Zebra Mussel Shell Fragments Found on the SW Inlet Side of CCW Heat Exchrs

Correspondence

10 CFR Part 50, Appendix R Exemption Request, Component Cooling Water Heat Exchanger and Pump Room, Fire Area T, Fire Zone T-1
Fire Protection - 10 CFR Part 50, Appendix R Exemption Requests (TAC Numbers 60994, 60995 and 61745)
Table 1 Conduit/Valve 1 Hour Fire Barrier Protection - CCW Heat Exchanger and Pump Room (Room 328)
Amendment No. 174 to Facility Operating License No. NPF-3 (TAC No. M822089)
IPE, Section 4, Flooding in the CCW Room
Unit No. 1 Construction Section XI, Technical Specifications for Heating, Ventilation, and Air Conditioning
Design Report for Mod 88-0055, Install Wafer Check Valves in Floor Drains
Fire Protection - Changes from Previous Submittals in Fire Protection Compliance Approaches, Serial Number 1757, 02/16/90
Fire Hazards Analysis, Fire Area Evaluation, Fire Area T
Generic Letter 86-10, Implementation of Fire Protection Requirements
Westinghouse letter to Toledo Edison, Operating Time of Type DHP breakers, dated 12/22/83
SE by the Office of NRR Supporting Exemption From Certain Requirements of Appendix R to 10 CFR Part 50, 11/23/82
Letter to NRC from Centerior Energy, Individual Plant Examination of External Events for Severe Accident Vulnerabilities, dated 12/16/96

Drawings

12501-M-1800-13-15, Sheet 1 of 2, Schematic Diagram Engine Control
12501-M-1800-13-15, Sheet 2 of 2, Schematic Diagram Engine Control
E-50B Sh 3A and Sh 4B, Revision 18, 4/24/98 Component Cooling Water System Component Cooling Pump 2 (AD113), 4.0kV
E-50B Sh 4A and Sh 4B , Revision 18, 4/24/98 Component Cooling Water System Component Cooling Pump 3 (AD108), 4.0kV
Motor Control Center circuit for Cooling Water System CC in Isolation Valve to CRD, ref. dwg. E-50B Sh 7A and Sh 7B , Revision 11, 1/11/92
Motor Control Center E11B, Circuit for Cooling Water System CC out Isolation Valve (inside containment), Ref. Dwg. E-50B Sh 9A and Sh 9B , Revision 9, 4/11/92
Motor Control Center E11B, Circuit for Cooling Water System CC out Isolation Valve (outside containment), Ref. Dwg. E-50B Sh 10A and Sh 10B , Revision 9, 4/11/92
FSK-M-HBC-35-4, Test Line Between 20" CCW Header and 10" Service Water Header, Revision3

FSK-M-HBC-35-5, Test Line Between 20" CCW Header and 10" Service Water Header, Revision1
OS-002, Makeup and Purification System, Revision 18

Vendor Drawings

A-2370, D-10 and D-20 Safety Relief Valve for Nuclear Service, Revision D
C-848, Component Cooling Surge Tank, Revision D, dated 04/15/97
V-831L-1A, 150# ANSI Three-Way Valve W/ Reverse Acting Diaphram Actuator & Top MTD Handwheel, Revision 1

Piping and Instrumentation Drawings (P&IDs)

M-036A, Component Cooling Water System, Revision 23
M-036B, Component Cooling Water System, Revision 27
M-036C, Component Cooling Water System, Revision 24
M-041A, Service Water Pumps and Secondary Service Water System, Revision 21
M-041B, Primary Service Water System, Revision 54

Licensee Event Reports

LER No. 346-98011, Manual Trip Due to Component Cooling Water System, 10/14/98,
LER No. 346-88016, Electrical Circuit Bridging in Safety Related Systems, 07/13/88,

Meeting Minutes

Maintenance Rule Expert Panel Meeting, 06/08/00

Memoranda

DBE-97-00357, Seismic Functions of Flow Switches (07/28/97)
DBE-97-00349, Modification 96-0005 Design Input (07/24/97)

Modifications

86-0251, Replace CCW Stop Check Valves, approved 07/07/86
87-0026, Change Instrument Set points for Fan C75-1
89-0013, Provide Fusing for Circuit 1CAC108K, 1CD1P23A, B, C, D, and 2CD2P23A, B, C, D.
95-0021, Ground Fault Relay
96-0005, Delete CCW Pump Low Flow and High Temperature Trips
97-0029, Fail Valves CC1471 and CC1474 in Open Position
98-0050, Provide a Different Means of Overpressure Protection for the Letdown Coolers
98-0056, Reduce Time Delay for Closing Non-Essential CCW Header Isolation Valves

Probability Risk Assessment

PSA System Notebook for 4160 v System, Revision 2, 10/26/99
PSA System Notebook, Component Cooling Water System (CCW), Revision 2, 10/27/99

Procedures

DB-FP-04024, Periodic Test Procedure, 18 Month Fire Damper Visual Inspection, Revision 4
DB-MS-01637, Scaffolding Erection and Removal Procedure, Revision 4
DB-OP-02001, R03, Electrical Distribution Alarm Panel 1, 11/4/99
DB-OP-02006, Reactor Coolant Pump Alarm Panel 6 Annunciators, Revision 2
DB-OP-02011, Heat Sink Alarm Panel 11 Annunciators, Revision 2
DB-OP-02501, Abnormal Procedure, Serious Station Fire, Revision 2
DB-OP-02523, Component Cooling Water System Malfunctions, Revision 1
DB-OP-06262-F02, Component Cooling Water System Procedure, 5/13/00, R02/C4
DB-PF-03070, Component Cooling Water Train 2 Quarterly Valve Testing, Revision 3
DB-PF-03071, Component Cooling Water Train 1 Quarterly Valve Testing, Revision 3
DB-PF-03100, Component Cooling Water Valve Refueling Test, Revision 2
DB-PF-03811, Miscellaneous Valves Quarterly Test, Revision 4
DB-PF-04151, Augmented CCW Integrated Leakage and Check Valves Leakage Test, Revision 0
NG-NA-00702, Corrective Action Program, Revision 3

Safety Evaluations

SE 97-0025/R01, Delete CCW Pump Low Flow and High Temperature Trips
SE 97-0025, Mod 96-0005, “Delete CCW Pump Low Flow and High Temperature Trips,” Revision 1
SE 96-0055, MOD 95-0056, UCN 96-205F, UCN 97-085F, UCN 97-117F, and UCN 98-041F Resolution to Thermo-Lag Fire Barrier Deficiencies, Revision 8
SE 97-0014, LM95-0021, Replace ITH Relays, dated 5/2/97
SE 97-0014, LM95-0021, Replace ITH Relays, Revision 1, dated 4/29/98

Specifications

Specification No. 7749-E-37, Technical Specification for Large Electric Motors for The Toledo Edison Company and The Cleveland Electric Illuminating Company Davis-Besse Nuclear Power Station Unit No. 1, Paragraph 10.6.3, Revision 4, 2/2/72

Standards and Codes

NFPA 13 - 1978, Standard for the Installation of Sprinkler Systems, 1978 Revision

System Description

System Description for CCW, SD-16, Revision 3, 12/21/99
System Description for Electrical Systems, SD-003A, Revision 3

Technical Specifications

Section 3/4.7.3, Component Cooling Water System

Test Reports

DB-SS-04001/R01	Component Cooling Water Pump Room Ventilation – Train 1
DB-SS-04002/R01	Component Cooling Water Pump Room Ventilation – Train 2
PT-5147.04.00	Component Cooling Water Pump Room Ventilation – Train 1
PT-5147.05.00	Component Cooling Water Pump Room Ventilation – Train 2
DB-SP-03032/R01	Service Water Pump-3 Refueling Test
DB-SP-03024/R01	Service Water Pump-2 Refueling Test
DB-SP-03018/R01	Service Water Pump-1 Refueling Test
DB-PF-03071/R03	Component Cooling Water Train 1 Quarterly Valve Testing
DB-PF-03070/R03	Component Cooling Water Train 2 Quarterly Valve Testing
DB-PF-03027/R03	Service Water Train 2 Quarterly Valve Test
DB-PF-03020/R03	Service Water Train 1 Quarterly Valve Test
DB-PF-05010/R03	Electrical Circuit Functional Test

Work Order 99-006410-001 – Post Modification Test for Modification 98-0056
Work Order 2-96-0005-001 – Post Modification Test for Modification 96-0005
Work Order 2-96-0005-002 – Post Modification Test for Modification 98-0005

Updated Final Safety Analysis Report Sections

Section 8, Electrical System
Section 6.3, Emergency Core Cooling System
Section 9.2.2, Component Cooling Water System